APPENDIX A

SCRAP METAL INVENTORIES AT U.S. NUCLEAR POWER PLANTS

Contents

Contents (continued)

page

Tables

page

Tables (continued)

page

Tables (continued)

page

Figures

SCRAP METAL INVENTORIES AT U.S. NUCLEAR POWER PLANTS

A.1 INTRODUCTION

At the end of 1999 the U.S. commercial nuclear power industry was represented by 104 operating reactors and 27 nuclear power reactors formerly licensed to operate (U.S. NRC 2000). In the next three decades, most of the operating licenses of reactors currently in operation—originally valid for 40 years—will have expired.[1]

With the publication of the NRC's Decommissioning Rule in June 1988 (U.S. NRC 1988), owners and/or operators of licensed nuclear power plants are required to prepare and submit plans and cost estimates for decommissioning their facilities to the NRC for review. Decommissioning, as defined in the rule, means to remove nuclear facilities safely from service and to reduce radioactive contamination to a level that permits release of the property for unrestricted use and termination of the license. The decommissioning rule applies to the site, buildings, and contents and equipment. Currently, several utilities have submitted a decommissioning plan to the NRC for review.

Historically, the NRC has defined three classifications for decommissioning of nuclear facilities:

- **DECON** is defined by the NRC as "the alternative in which the equipment, structures, and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations."

- **SAFSTOR** is defined as "the alternative in which the nuclear facility is placed and maintained in a condition that allows the nuclear facility to be safely stored and subsequently decontaminated (deferred dismantlement) to levels that permit release for unrestricted use."

  The SAFSTOR decommissioning alternative provides a condition that ensures public health and safety from residual radioactive contamination remaining at the site, without the need for extensive modification to the facility. Systems not required to be operational for fuel storage, maintenance and surveillance purposes during the dormancy period are to be drained, de-energized and secured.

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[1] As stated in Chapter 2, the NRC has issued a rule allowing a licensee to apply for a 20-year renewal of its original operating license. To date, five reactors have been granted such license renewals; a number of other renewal applications are pending, and more applications are anticipated.

- **ENTOMB** is defined as "the alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete; the entombed structure is appropriately maintained and continued surveillance is carried out until the radioactive material decays to a level permitting unrestricted release of the property."

Over the years, the basic concept of the three alternatives has remained unchanged. However, because of the accumulated inventory of spent nuclear fuel (SNF) in the reactor storage pool and the requirement for about seven years of pool storage for the SNF before transfer to dry storage, the timing and steps in the process for each alternative have had to be adjusted to reflect present conditions. For the DECON alternative, it is assumed that the owner has a strong incentive to decontaminate and dismantle the retired reactor facility as promptly as possible, thus necessitating transfer of the stored SNF from the pool to a dry storage facility on the reactor site. While continued storage of SNF in the pool is acceptable, the 10 CFR Part 50 license could not be terminated until the pool had been emptied, and only limited amounts of decontamination and dismantlement of the facility would be required. This option also assumes that an acceptable dry transfer system will be available to remove the SNF from the dry storage facility and to place it into licensed transport casks when the time comes for DOE to accept the SNF for disposal at a high level waste repository.

In addition, the amended regulation stipulates that alternatives, which significantly delay completion of decommissioning, such as use of a storage period, will be acceptable if sufficient benefit results. The Commission indicated that a storage period of up to 50 years and a total of 60 years between shutdown and decommissioning is a reasonable option for decommissioning a light water reactor. In selecting 60 years as an acceptable period of time for decommissioning of a nuclear power reactor, the Commission considered the amount of radioactive decay likely to occur during an approximately 50-year storage period and the time required to dismantle the facility.

In summary, the reactor facility will need to adequately cool the high-burnup assemblies from the final fuel core in the pool for up to seven years and must fulfill the regulatory requirements that critical support systems be maintained in operable conditions. Therefore, the time between shutdown, decontamination and the earliest date of dismantling efforts that would generate scrap metal is likely to be about 10 years. This interval may extend up to 60 years under the SAFSTOR decommissioning alternative. A longer time interval has the obvious benefit of greatly reducing radionuclide inventories through radioactive decay. However, a simple inverse correlation between reduced levels of contamination and increased quantities of scrap metal with a potential for clearance cannot be inferred. It is likely that for most scrap metal, the longer decay time may merely affect the choice of decontamination method and/or decontamination effort required to meet a desired standard. For example, a storage period that reduces beta/gamma surface contamination of 10^7 dpm/100 cm^2 at 10 years post-shutdown to 10^5 dpm/100 cm^2 (i.e., a 100-fold reduction) would still require substantial decontamination in order to meet current standards defined by NRC Regulatory Guide 1.86 (U.S. AEC 1974). However, since the reduced activity would most likely be dominated by Cs-137, the method and level of effort required for successful decontamination would be different than that employed at an earlier time.

The potential for clearance of scrap metal is, therefore, dictated by the cost-effectiveness with which materials can be decontaminated to acceptable levels. Estimates of scrap metal quantities must consider starting levels of contamination and whether the contamination is surficial or volumetrically distributed.

Residual radioactive contaminants of reactor components/systems and building structures is generally grouped as: (1) activation products that are distributed volumetrically, (2) activation and fission products in the form of corrosion films deposited on internal surfaces, and (3) contamination of external surfaces that result from the deposition of liquid and airborne radioactive materials associated with steam, reactor coolant and radioactive waste streams.

Most of the scrap metal generated by the complete dismantling of a nuclear power plant is not expected to be radioactive. The non-radioactive scrap includes the large quantities of structural metals and support systems that have *not* been exposed to radioactivity during reactor operations. Conversely, some metal components will undoubtedly be so contaminated as to render them unsuitable for clearance.

A.2 CHARACTERISTICS OF REFERENCE REACTOR FACILITIES

A crucial factor affecting the quantity of metal and associated contamination levels is the basic design of the reactor. Each of the nuclear power reactors currently operating in the U.S. is either a pressurized water reactor (PWR) or a boiling water reactor (BWR). Of the 104 operating reactors, 35 are BWRs manufactured by General Electric and 69 are PWRs manufactured by Westinghouse, Combustion Engineering and Babcock and Wilcox (U.S. NRC 2000).

Appendix A-1 provides a complete listing of U.S. nuclear power reactors along with demographic data that includes projected year of shutdown.

In the 1976-1980 time frame, two studies were carried out for the NRC by the Pacific Northwest Laboratory (PNL) that examined the technology, safety and costs of decommissioning large reference nuclear power plants. Those studies—"Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," NUREG/CR-0130 (Smith et al. 1978) and "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station," NUREG/CR-0672 (Oak et al. 1980)—reflected the industrial and regulatory situation of the time.

To support the final Decommissioning Rule issued in 1988, the earlier PNL studies were updated with the issuance of "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Station," NUREG/CR-5884 (Konzek et al. 1995) and "Revised Analyses of Decommissioning for the Reference Boiling Water Reactor Power Station," NUREG/CR-6174 (Smith et al. 1996). The four NUREG reports cited above, along with several other NRC reports and selected decommissioning plans on file with the Commission, represent the primary source of information used to characterize Reference PWR and BWR facilities and to derive estimates of scrap metal inventories for the industry at large.

A.2.1 Reference PWR Design and Building Structures

The Reference PWR facility is the 3,500 MWt (1,175 MWe) Trojan Nuclear Plant (TNP) at Rainier, Oregon, operated by the Portland General Electric Company (PGE). Designed by Westinghouse, this reactor is considered a typical PWR that has been cited as the Reference PWR (Smith et al. 1978; Konzek et al. 1995).

The NRC granted the operating license for the TNP on November 21, 1975, and the plant formally began commercial operation on March 20, 1976. TNP's operating license was scheduled to expire on February 8, 2011. However, on November 9, 1992, the TNP was shut down when a leak in the "B" steam generator was detected and the licensee notified the NRC of its decision to permanently cease operations in January 1993. Following the transfer of spent fuel from the reactor vessel to the spent fuel pool in May of 1993, TNP's operating license was reduced to a possession only license. TNP's 17-year operating period encompassed 14 fuel cycles and approximately 3,300 effective full-power days. In the decommissioning plan submitted by PGE, the licensee has proposed the DECON approach with a five-year delay period prior to decontamination and dismantlement (Portland General Electric 1996).

In a PWR, the primary coolant is heated by the nuclear fuel core but is prevented from boiling by a pressurizer, which maintains a pressure of about 2,000 psi. The principal systems and components of the nuclear steam supply system are illustrated in Figure A-1. Components of interest are the reactor vessel, which contains the fuel and coolant, and the reactor coolant system (RCS). The reactor vessel also contains internal support structures (not shown) that constrain the fuel assemblies, direct coolant flow, guide in-core instrumentation and provide some neutron shielding. The RCS consists of four loops for transferring heat from the reactor's primary coolant to the secondary coolant system. Each loop consists of a steam generator, a reactor coolant pump and connecting piping. Steam generated from secondary feedwater is passed through the turbine, condensed back to water by the condenser and recycled.

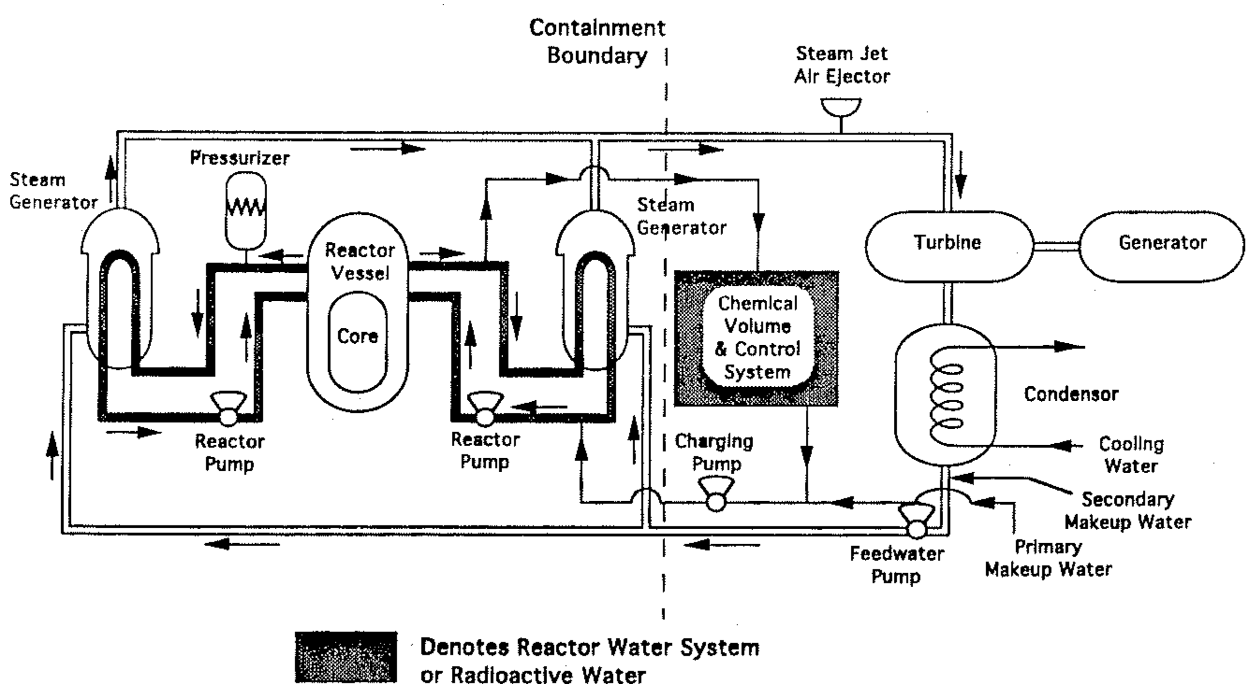

Figure A-1. Pressurized Water Reactor (Dyer 1994)

Also included in the primary loop is a small side-stream of water that is directed to the chemical volume and control system (CVCS). The CVCS provides chemical and radioactive cleanup of the primary coolant through demineralizers and evaporators. The primary coolant is reduced in both pressure and temperature by the CVCS before being processed; therefore, the CVCS is often referred to as the letdown system. The water processed through the CVCS is returned to the primary loops by the charging pumps. Note that the primary coolant processed through the CVCS is brought through the containment boundary or out of the containment building, but the primary coolant providing the heat transfer to the steam generators does not pass through the containment boundary.

As shown in Figure A-1, highly contaminated components of a PWR are those associated with the primary coolant system. Low-level contamination of the secondary loop is a result of steam generator tube leakage in which limited quantities of primary coolant are introduced into the recirculating steam/water. Other major contaminated systems of PWRs not shown in Figure A-1 include the radioactive waste handling system and the spent fuel storage system.

The principal structures requiring decontamination for license termination at the Reference PWR are the (1) reactor building, (2) fuel building and (3) auxiliary building. In addition to housing major plant systems, all three buildings contain contaminated systems and substantial quantities of contaminated structural metals that are candidates for clearance.

A.2.1.1 Reactor Building

The reactor building houses the nuclear steam supply system. Since its primary purpose is to provide a leak-tight enclosure under normal as well as accident conditions, it is frequently referred to as the containment building. Major interior structures include the biological shield, pressurizer cubicles and a steel-lined refueling cavity. Supports for equipment, operating decks, access stairways, grates and platforms are also part of the containment structure internals.

The reactor building is in the shape of a right circular cylinder, approximately 64 m tall and 22.5 m in diameter. It has a hemispherical dome, a flat base slab with a central cavity and an instrumentation tunnel.

A.2.1.2 Fuel Building

The fuel building—approximately 27 m tall, 54 m long, and 19 m wide—is a steel and reinforced concrete structure with four floors. This building contains the spent-fuel storage pool and its cooling system, much of the CVCS, and the solid radioactive waste handling equipment. Major steel structural components include fuel storage racks and liner, support structures for fuel handling, and components, ducts and piping associated with air conditioning, heating, cooling and ventilation.

A.2.1.3 Auxiliary Building

The auxiliary building—approximately 30 m tall, 35 m long and 19 m wide—is a steel and reinforced concrete structure with two floors below grade and four floors above grade. Principal systems contained in the auxiliary building include the liquid radioactive waste treatment systems, filter and ion exchanger vaults, waste gas treatment system, and the ventilation equipment for the containment, fuel and auxiliary buildings.

A.2.1.4 Control and Turbine Buildings

Other major building structures with substantial metal inventories include the control building and the turbine building. The principal contents of the control building are the reactor control room, and process and personnel facilities. The principal systems contained in the turbine building are the turbine generator, condensers, associated power production equipment, steam generator auxiliary pumps, and emergency diesel generator units.

Barring major system failures (e.g., steam generator failure) most scrap metal derived from these systems can be assumed to be free of contamination and can, therefore, be excluded from the inventories of scrap metal which are candidates for clearance.

A.2.2 Reference BWR Design and Building Structures

The 3,320 MWt (1,155 MWe) Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 located near Richland, Wash., is the basis for the Reference BWR facility (Oak et al. 1980; Smith et al. 1996).

The design of a BWR (see Figure A-2) is simpler than a PWR inasmuch as the reactor coolant water is maintained near atmospheric pressure and boiled to generate steam. This allows the coolant to directly drive the turbine. Thereafter, the steam is cooled in the condenser and returned to the reactor vessel to repeat the cycle. In a BWR, the contaminated reactor coolant comes in contact with most major reactor components, including the reactor vessel and piping, steam turbine, steam condenser, feedwater system, reactor coolant cleanup system and steam jet

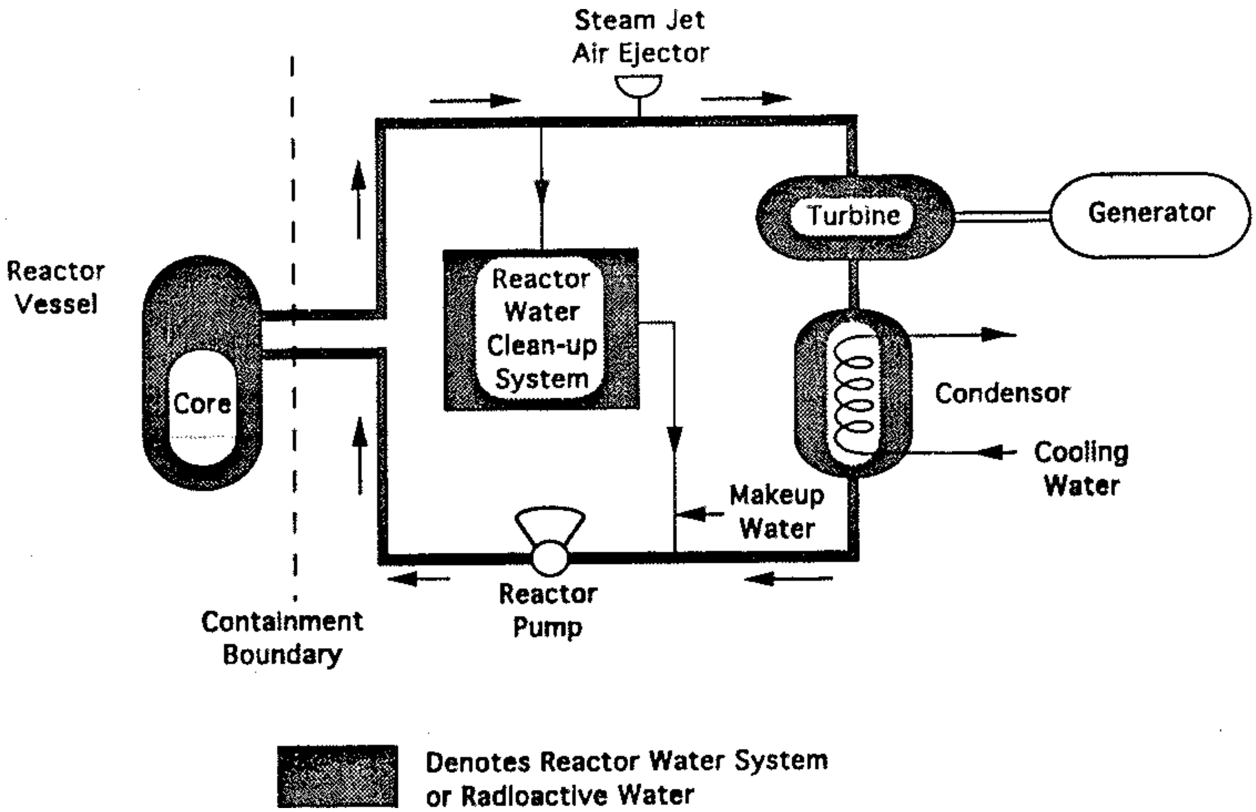

Figure A-2. Boiling Water Reactor (Dyer 1994)

air ejector system. As with the PWR, other major contaminated systems include the radioactive waste treatment system and spent fuel storage system.

The principal buildings requiring decontamination and dismantlement in order to obtain license termination at the reference BWR power station are the reactor building, the turbine generator building, and the radwaste and control building. These three buildings contain essentially all of the activated or radioactively contaminated material and equipment within the plant.

A.2.2.1 Reactor Building

The reactor building contains the nuclear steam supply system and its supporting systems. It is constructed of reinforced concrete capped by metal siding and roofing supported by structural steel. The building surrounds the primary containment vessel, which is a free-standing steel pressure vessel. The exterior dimensions of the Reactor Building are approximately 42 m by 53 m in plan, 70 m above grade and 10.6 m below grade to the bottom of the foundation.

A.2.2.2 Turbine Building

The turbine building, which contains the power conversion system equipment and supporting systems, is constructed of reinformed concrete capped by steel-supported metal siding and roofing. This structure is approximately 60 m by 90 m in plan and 42.5 m high.

A.2.2.3 Radwaste and Control Building

The radwaste and control building houses, among other systems: the condenser off-gas treatment system, the radioactive liquid and solid waste systems, the condensate demineralizer system, the reactor coolant cleanup demineralizer system and the fuel-pool cooling and cleanup demineralizer system. The building is constructed of reinforced concrete, structural steel, and metal siding and roofing. This structure is approximately 64 by 49 m in plan, 32 m in overall height, and stands as two full floors and one partial floor above the ground floor.

A.3 RESIDUAL ACTIVITIES IN REFERENCE REACTOR FACILITIES

Significant levels of contamination remain in a nuclear power station following reactor shutdown, even after all spent nuclear fuel has been removed. Neutron-activated structural materials in and around the reactor pressure vessel contain most of the residual activity in a relatively immobile condition. Other sources of radioactive contamination comprise activated corrosion products and fission products leaked from failed fuel, which are transported throughout the station by the reactor coolant streams. The origin and mobility of radioactive contaminants following reactor shutdown leads to grouping of residual activities into five categories of different binding matrices. These categories include:

1. **Activated Stainless Steel**. Reactor internals, composed of Type 304 stainless steel, become activated by neutrons from the core. Radionuclides have very high specific activities and are immobilized inside the corrosion-resistant metal.

2. **Activated Carbon Steel**. Reactor pressure vessels are made of SA533 carbon steel that becomes activated by neutron bombardment. The specific activities are considerably lower than in the stainless steel internals, and the binding matrix is much less corrosion resistant.

3. **Activated Structural Steel, Steel Rebar and Concrete**. In the reactor cavity, these components become activated by neutrons escaping from the reactor vessel. Significant

activation occurs along approximately 15 feet of the reactor cavity vertically centered on the reactor core and to a depth of about 16 inches in the concrete.

4. **Contaminated Internal Surfaces of Piping and Equipment**. Activated corrosion and fission products travel through the radioactive liquid systems in the plant. A portion forms a hard metallic oxide scale on the inside surfaces of pipes and equipment.

5. **Contaminated External Surfaces**. External surfaces may become contaminated over the lifetime of the plant, primarily from leaks, spills and airborne migration of radionuclides contained in the reactor coolant water (RCW). The specific activity of RCW is low, but the contamination is easily mobilized and may be widespread.

All of the neutron-activated metals/materials are contained in the reactor pressure vessel, vessel internals, and structural components inside and within the concrete biological shield.

Total quantities and the relative radionuclide composition of the residual activity are not only affected by reactor design (BWR vs. PWR) but are also strongly influenced by numerous other factors including (1) fuel integrity, (2) rated generating capacity and total years of operation, (3) composition of metal alloys in reactor components and the RCS, (4) coolant chemistry and water control measures, and (5) the performance and/or failures of critical systems and their maintenance over the initial 40-year span of the operating license (see footnote on page A-1).

Table A-1 provides summary estimates of typical residual activities for each of the five major source categories. Inspection of the data reveals that the volumetrically activated stainless steel represents the overwhelming majority of the residual activities. Much smaller activities are found in volumetrically activated carbon steel and internal and external surface contamination consisting of activation and fission products. A more detailed discussion of residual activity by source category is given below.

A.3.1 Neutron-Activated Reactor Components and Structural Materials

Contamination of reactor components and structural materials by neutron activation is the result of normal reactor operation. The interaction of neutrons with constituents of stainless steel, carbon steel and concrete in and around the reactor vessel results in high in-situ activities. The radionuclide inventories include significant activities of Cr-51, Mn-54, Fe-55, Fe-59, Co-58, Ni-59 and Ni-63. The specific activities of various radionuclides in materials exposed to a neutron flux is highly variable and depends upon (1) the concentration of the parent nuclide and its neutron cross-section, (2) the radioactive half-life of the radionuclide, (3) the neutron flux intensity at the given location, and (4) the duration of neutron exposure.

Table A-1. Sources of Residual Activities in Reference BWR and PWR

Source	Residual Activity (Ci)	
	BWR[a]	PWR[b]
Activated Stainless Steel	6.6e+06	4.8e+06
Activated Carbon Steel	2.9e+03	2.4e+03
Activated Structural Components, Rebar, Metal Plates, I-Beams	1.2e+03	1.2e+03
Internal Surface Contamination of Piping and Equipment	8.5e+03	4.8e+03
External Contamination of Equipment, Floors, Walls, Other Surfaces	1.1e+02	1.1e+02 [c]

[a] Oak et al. 1980
[b] Smith et al. 1978
[c] Implied value (U.S. NRC 1994)

A.3.1.1 Reference BWR

The average activity concentrations and estimated total activities for Reference BWR structural components with significant amounts of neutron activation are listed in Table A-2.

The Reference BWR reactor vessel is fabricated of SA533 carbon steel about 171 mm thick and is clad internally with 3 mm of Type 304 stainless steel. The total mass of the empty vessel is about 750 metric tons (t). The major internal components include the fuel core support structure; steam separators and dryers; coolant recirculation jet pumps; control rod guide tubes; distribution piping for feedwater, core sprays and liquid control; in-core instrumentation, and miscellaneous other components. Collectively, these internals, made of stainless steel, represent about 250 t.

A.3.1.2 Reference PWR

The right circular cylinder of the Reference PWR is constructed of carbon steel about 216 mm in thickness and is clad on the inside with stainless steel or Inconel having a thickness of about 4 mm. The approximate dimensions of the vessel are 12.6 m high and 4.6 m in outer diameter. The vessel weighs about 400 t.

Table A-2. Estimated Activities of Neutron-Activated Reactor Components in a BWR

Component (number)	Average Activity Concentration (Ci/m^3)	Total Activity (Ci)
Core Shroud (1)	1.68e+06	6.30e+06
Jet Pump Assembly (10)	2.62e+04	2.00e+03
Reactor Vessel (1)		2.16e+03
Cladding	1.07e+03	
Shell Wall	1.12e+02	
Steam Separator Assembly (1)		9.60e+03
Shroud Head Plant	1.03e+04	
Steam Separator Risers	2.53e+03	
Top Fuel Guide (1)	9.71e+04	3.01e+04
Orificed Fuel Support (193)	1.01e+03	7.01e+02
Core Support Plate (1)	2.56e+02	6.50e+02
Incore Instrument Strings (55)	7.67e+05	1.10e+04
Control Rod (185)	5.11e+05	1.78e+05
Control Rod Guide Tube (185)	2.16e+02	9.47e+01
Total		6.55e+06

Source: Oak et al. 1980

The vessel's internal structures support and constrain the fuel assemblies, direct coolant flow, guide in-core instrumentation and provide some neutron shielding. The principal components are: the lower core support assembly, which includes the core barrel and shroud, with neutron shield pads, and the lower core plate and supporting structure; and the upper core support and in-core instrumentation support assemblies. These structures are made of 304 stainless steel and have a total mass of about 190 t.

Based on 40 years of facility operation and assuming 30 effective full-power years (EFPY) of reactor operation, the total activity contained in the activated vessel and internals is estimated to be 4.8 million curies (see Table A-3). Extra-vessel materials subject to significant neutron activation ($\approx$10 curies) includes the reactor cavity steel liner and a limited quantity of reinforcement steel (rebar). Additionally, the concrete bioshield contains an estimated total inventory of about 1,200 curies.

Table A-3. Neutron-Activated Reactor Components in a PWR

Component	Average Activity Concentration (Ci/m^3)	Total Activity (Ci)
Shroud	2.97e+06	3.43e+06
Lower 4.7 m of core barrel	3.07e+05	6.52e+05
Thermal shield	1.45e+05	1.46e+05
Vessel inner cladding	7.73e+03	1.50e+03
Lower 5.02 m of vessel wall	9.04e+02	1.76e+04
Upper grid plate	4.20e+04	2.43e+04
Lower grid plate	1.12e+06	5.53e+05
Total		4.82e+06

Source: Smith et al. 1978

The projected estimates of Table A-3 for the Reference PWR (i.e., Trojan Nuclear Plant) made in 1978 can be compared to the more current estimates contained in that plant's decommissioning plan (submitted to the NRC in 1996). Table A-4 identifies revised calculated inventories of activation products for 1993, or one year after shutdown. The recalculated value of about 4.2 million curies is about 13% lower than the original estimate of 4.8 million curies and principally reflects the difference between 17 years of actual plant operation and the initial projection of 40 years.

Table A-4. Activation Levels at Trojan Nuclear Plant One Year after Shutdown

System	Activity (Ci)
Reactor Vessel	6.20e+03
Reactor Vessel Internals	4.16e+06
Vessel Clad and Insulation	2.37e+04
Bioshield Wall	8.30e+02
Total	4.19e+06

The considerably higher activities calculated for a Reference BWR primarily reflect the larger size and mass of the vessel and its internals.

For both PWR and BWR plants, the range of activity concentrations among individual reactor components at time of shutdown is likely to vary over several orders of magnitude. Nevertheless, even those components with the lowest activity concentrations would still have residual activities far in excess of any conceivable levels that would permit clearance. (Note: at a specific gravity of 7.86, a cubic meter of steel containing one curie has a specific activity of 0.13 μCi/g.) Furthermore, these components also exhibit high levels of interior surface contamination. While surface contamination is potentially removable, the volumetrically distributed activation products are not.

For this reason, the reactor vessel and all internal components identified in Tables A-2 and A-3 must be excluded from plant material inventories which are potential candidates for clearance. Excluded for similar reasons are certain metal components used for structural support and reinforcement (i.e., rebar, I-beams, and floor and reactor cavity liner plates) that exhibit significant levels of activation products.

Scrap metal that can potentially be cleared can therefore originate only in reactor systems and structural components where contamination is limited to interior and exterior surfaces.

A.3.2 Internal Surface Contamination of Equipment and Piping

Activated corrosion products from structural materials in contact with the reactor coolant and fission products from leaking fuel contribute to the radioactive contamination of reactor coolant streams during plant operation. Although most of these contaminants are removed through filtration and demineralization by the CVCS, a small portion remains in the coolant. With time, some of the contaminants, principally the neutron-activated, insoluble corrosion products, tend to deposit on inner surfaces of equipment and piping systems. The resulting metal oxide layer consists primarily of iron, chromium and nickel with smaller, but radiologically significant, quantities of cobalt, manganese and zinc. This section characterizes the mixture of internal surface contaminants and their relative distribution within major components associated with BWR and PWR power plants.

A.3.2.1 Measurements of Internal Surface Contamination at Six Nuclear Power Plants

In a 1986 PNL study, six nuclear power plants—three PWRs and three BWRs—were assessed for residual inventories and distributions of long-lived radionuclides following plant shutdown (Abel et al. 1986). Residual concentrations in the various plant systems decreased in the following order: (1) primary coolant loop, (2) radwaste handling system, and (3) secondary coolant loop in PWRs and condensate system in BWRs. Table A-5 lists total estimated activities at the six plants, as well as the electrical ratings and the approximate number of operational years of the plants at the time of the assessments. The operational periods ranged from 8.3 years for Turkey Point Unit 3 to slightly over 18 years for Dresden Unit 1.

Table A-5. Residual Activities and Operating Parameters of Six Nuclear Power Plants*

Stations	Total Inventory (Ci)	Period of Operation (y)	Power Rating (MWe)	Reactor Type
Humboldt Bay	600	13	63	BWR
Dresden-1	2,350	18.3	210	BWR
Monticello	514	10	550	BWR
Indian Point-1	1,050	11	170	PWR
Turkey Point-3	2,580	8.3	660	PWR
Rancho Seco	4,470	8.8	935	PWR

Source: Abel et al. 1986

* Total inventory includes radionuclides with half-lives greater than 245 days (i.e., Zn-65); inventories in activated metal components of the reactor pressure vessel and internals and activated concrete are excluded.

The relative radionuclide composition of internally contaminated surfaces at the six plants also showed considerable variation (see Table A-6). Fluctuations in compositions were due to numerous factors including: (1) the elapsed time since reactor shutdown; (2) rated generating capacity; (3) materials of construction of the operating systems; (4) reactor type (PWR or BWR); (5) coolant chemistry and corrosion control; (6) fuel integrity during operations; and (7) episodic equipment failure and leakage of contaminated liquids.

Inventories include only the radioactive contamination of corrosion film and crud[2] on surfaces of the various plant systems, and do not include the highly activated components of the pressure vessel. The most abundant radionuclides in samples two to three months old included Mn-54, Fe-55, Co-58, Co-60 and Ni-63. Zinc-65 was present in relatively high concentrations in BWR corrosion film samples. However, Fe-55, and Co-57+Co-60 were the most abundant radionuclides at all stations except Monticello. These radionuclides constituted over 95% of the

[2] A colloquial term for corrosion and wear products (rust particles, etc.) that become radioactive (i.e., activated) when exposed to radiation. The term is actually an acronym for Chalk River Unidentified Deposits, the Canadian plant at which the activated deposits were first discovered.

estimated inventories at Humboldt Bay and Turkey Point. At Indian Point-1, Dresden-1, Turkey Point-3 and Rancho Seco, they accounted for 82, 74, 98 and 70%, respectively, of the total estimated inventory. Although Fe-55 and Co-60 accounted for the majority of the inventory (greater than 60% at five of the six stations), the relationship between the two radionuclides was quite variable. The transuranic nuclides (Pu-238, Pu-239, Pu-240, Am-241, Cm-242 and Cm-244) constituted varying percentages of the total inventory, ranging from 0.001% at Rancho Seco to 0.1% at Dresden-1.

Table A-6. Relative Activities of Long-Lived Radionuclides at Six Nuclear Power Plants*

Radionuclide	Relative Activity, Decay-Corrected to Shutdown Date (%)†					
	BWRs			PWRs		
	Humboldt Bay	Dresden-1	Monticello	Indian Point-1	Turkey Point-3	Rancho Seco
Mn-54	3	0.9	1	4	0.4	4
Fe-55	90	28	1	67	31	28
Co-57	—	—	—	—	43	24
Co-60	6	46	11	15	24	18
Ni-59	—	0.09	—	0.02	4e-03	0.1
Ni-63	0.2	5	0.04	2	0.1	19
Zn-65	—	19	84	11	1	0.09
Sr-90	4e-03	7e-03	2e-03	7e-04	8e-04	< 0.01
Nb-94	< 4e-03	< 3e-03	< 0.1	8e-04	< 4e-03	< 4e-03
Tc-99	3e-04	4e-05	8e-05	8e-05	8e-03	< 5e-03
Ag-110m	—	—	—	—	—	4
I-129	< 3e-06	< 1e-05	< 1e-06	2e-05	< 3e-03	< 1e-05
Cs-137	0.5	0.04	2	0.5	—	0.4
Ce-144	—	1	—	—	0.2	< 0.04
TRU**	5e-03	0.1	8e-03	2e-03	6e-03	1e-03
Total (Ci)	596	2,350	448	1,070	2,580	4,460

Source: Abel et al. 1986

* Excludes activated metal components of the reactor pressure vessel and internals and activated concrete.

† Relative activity of each nuclide as a percentage of total activity at each power plant

** Transuranic alpha-emitting radionuclides with half-lives greater than 5 years, including Pu-238, Pu-239, Pu-240, Am-241, Am-243 and Cm-244.

Secondary coolant loops in PWRs and condensate systems in BWRs contained much lower activity concentrations than observed in primary loop or feedwater samples. Typically, concentrations were two or more orders of magnitude lower in secondary system samples.

As expected, the steam generators contained the single largest repository of internally deposited radionuclides at the PWR stations examined (see Table A-7). The percentages of the total residual radionuclide inventories in the steam generators were 77, 89 and 94% for Indian Point-1, Turkey Point-3 and Rancho Seco, respectively. The other repository of significance in a PWR is the radwaste system, which typically contained 5 to 10% of the total residual inventory.

Table A-7. Distribution of Activities in Major Systems of Three PWRs (%)

System	Turkey Point-2	Indian Point-1	Rancho Seco	Average
Steam Generators	89	77	94	86.7
Pressurizer	0.5	0.5	0.33	0.4
RCS Piping	0.9	2.6	0.71	1.4
Piping (Except RCS)	< 0.01	14	< 0.01	4.7
Secondary Systems	0.1	0.2	0.05	0.1
Radwaste	9.2	7	5	7.1

Source: Abel et al. 1986

A.3.2.2 Internal Surface Contamination Levels Reported in Decommissioning Plans

A small number of commercial nuclear power facilities, which have experienced a premature shutdown or have projected shutdown within the next few years, have submitted a decommissioning plan to the NRC for review. Summarized below are system-specific internal contamination levels reported for one BWR and two PWRs.

Big Rock Point Nuclear Plant
The Big Rock Point Nuclear Plant is a small (67 MWe) BWR designed by the General Electric Company and constructed by Bechtel Power Corporation. Owned and operated by Consumers Power Company, the plant started commercial operation in March 1963 and was shut down in August 1997. Table A-8 presents summary data of systems internally contaminated (Consumers Power 1995).

Table A-8. Internal Contamination Levels of Big Point Nuclear Plant at Shutdown

System	Surface Contamination Level (dpm/100 cm^2)
Liquid Rad Waste Tanks	3e+10
Nuclear Steam Supply	9e+09
RDS	3e+09
Main Steam System	4e+08
Fuel Pool	4e+08
Liquid Radwaste System	4e+08
Condensate System	5e+07
Resin Transfer System	3e+07
Off-gas System	3e+07
Control Rod Drive	6e+06
Rad Waste Storage	9e+05
Fuel Handling Equip	7e+05
Heating & Cooling System	3e+05

San Onofre Nuclear Generation Station Unit 1 (SONGS 1)

SONGS 1 is a 436-MWe PWR that started operation in 1968. As a result of an agreement with the California Public Utility Commission, operation of SONGS 1 was permanently discontinued on November 30, 1992 at the end of fuel cycle #11. A preliminary decommissioning plan, submitted to the NRC on December 1, 1992, proposed to maintain SONGS 1 in safe storage until the permanent shutdown of SONGS 2 and 3. SONGS 2 and 3 are licensed to operate until 2013.

In support of the SONGS 1 decommissioning plan, scoping surveys and analyses were performed that supplemented an existing radiological data base (Southern California Edison 1994). The containment building, fuel storage building and radwaste/auxiliary building were identified as the principal structures containing significant levels of radioactivity within plant systems. Systems were grouped by contamination levels defined as (1) highly contaminated, (2) medium-level contaminated and (3) low-level contaminated. Based on total radionuclide inventories and surface areas, an average contamination level for each of the three groupings was derived (see Table A-9).

Table A-9. Plant Systems Radioactivity Levels at SONGS 1

Plant Systems	Total Area (cm^2)	Surface Contamination Level (dpm/100 cm^2)	Total Activity (Ci)
High-Level Contaminated Systems:	1.26e+08	3.6e+09	2.08e+03
LDS Letdown			
PAS Post Accident Sampling System			
PZR Pressurizer Relief			
RCS Reactor Coolant			
RHR Residual Heat Removal			
RSS Reactor Sampling			
SFP Spent Fuel Pool Cooling			
VCC Volume Control			
Medium-Level Contaminated Systems:	1.25e+08	1.9e+06	1.08e+01
BAS Boric Acid			
CWL Containment Water Level			
RCP RCP Seal Water			
RLC Radwaste Collection			
RMS Radiation Monitoring			
RWG Radwaste Gas			
RWL Radwaste Liquid			
CRS (Containment Spray) Recirculation			
SIS Safety Injection			
Low-Level Contaminated Systems:	3.18e+08	8.3e+03	1.21e-02
AFW Auxiliary Feedwater			
CCW Component Cooling			
CND Condensate			
SHA Sphere Hydrazine Addition			
CSS Condensate Sampling			
CVD Condensate Vents & Drains			
CVI Cryogenics			
CWS Circulating Water			
FES Flash Evaporator			
FPS Fire Protection			
FSS Feed Sampling			
FWH Feedwater Heaters			
FWS Feedwater			
MSS Main Steam			
MVS Miscellaneous Ventilation			
PSC Turbine Sample Cooling			
SDW Service Water			
SWC Salt Water Cooling			
TCW Turbine Cooling			

Continue